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NL-13-113

August 27, 2013

U.S. Nuclear Regulatory Commission
Document Control Desk
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Rockville, MD 20852-2738

SUBJECT: Licensee Event Report # 2013-003-00, "Manual Reactor Trip Due to Decreasing Steam Generator Water Levels Due to Loss of Main Feedwater (FW) Flow Caused by a Loss of Instrument Air to the FW Regulating Valves"
Indian Point Unit No. 2
Docket No. 50-247
DPR-26

Dear Sir or Madam:

Pursuant to 10 CFR 50.73(a)(1), Entergy Nuclear Operations Inc. (ENO) hereby provides Licensee Event Report (LER) 2013-003-00. The attached LER identifies an event where the reactor was manually tripped, which is reportable under 10 CFR 50.73(a)(2)(iv)(A). As a result of the reactor trip, the Auxiliary Feedwater System was actuated and two Main Steam Isolation Valves closed, which is also reportable under 10 CFR 50.73(a)(2)(iv)(A). This condition was recorded in the Entergy Corrective Action Program as Condition Report CR-IP2-2013-02717.

There are no new commitments identified in this letter. Should you have any questions regarding this submittal, please contact Mr. Robert Walpole, Manager, Licensing at (914) 254-6710.

Sincerely,

JAV/cbr

cc: Mr. William Dean, Regional Administrator, NRC Region I
NRC Resident Inspector's Office
Ms. Bridget Frymire, New York State Public Service Commission

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NRR

NRC FORM 366 (9-2007) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104 EXPIRES: 10/31/2013

LICENSEE EVENT REPORT (LER)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
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4. TITLE: Manual Reactor Trip Due to Decreasing Steam Generator Water Levels Due to Loss of Main Feedwater (FW) Flow Caused by a Loss of Instrument Air to the FW Regulating Valves

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV. NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
07	03	2013	2013-	003 -	00	08	27	2013		05000
									FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE: 1

10. POWER LEVEL: 100%

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☒ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
Robert Altadonna, Codes Engineer, Program & Codes Engineering	(914) 254-7242

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
B	LD	CPLG	W120	Y					

14. SUPPLEMENTAL REPORT EXPECTED		15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)*	☒ NO				

16. ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced type written lines)*

On July 3, 2013, operators initiated a manual reactor trip as a result of lowering steam generator (SG) levels due to the loss of feedwater (FW) from the trip of both main FW pumps. All control rods fully inserted and all required safety systems functioned properly. The plant was stabilized in hot standby with decay heat being removed by the main condenser. The Auxiliary Feedwater System automatically started as expected. Investigations determined the decreasing SG levels were due to a loss of main FW flow as a result of the closure of the FW regulating valves. The FW regulating valves closed due to a loss of instrument air (IA) pressure. The IA pressure was lost when a two inch copper IA tubing in the 22 Main Transformer moat separated at a soldered coupling. Prior to the event piping lines including the IA line buried in the main transformer moat were excavated and temporary supports installed. The apparent cause was poor legacy workmanship assembling the IA tubing coupling during original plant construction. The IA tubing was not fully inserted into the coupling resulting in reduced joint strength. Corrective actions included reassembly and soldering of the IA joint with full insertion, acoustic emission and snoop testing on repaired coupling. Axial and thrust restraints were installed on the IA line in the moat. A caution was placed in the Buried Piping Program database associated with buried copper tubing identifying the potential for the separation of soldered joints when the line is excavated and the need for restraints or other contingencies to minimize the probability of a line separation. The event had no effect on public health and safety.

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NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

Note: The Energy Industry Identification System Codes are identified within the brackets {}.

DESCRIPTION OF EVENT

On July 3, 2013, while at 100% steady state reactor power, Control Room operators received a Steam Flow/FW Flow Mismatch alarm and entered procedure 2-AOP-FW-1 when both main boiler FW pumps tripped. Operators initiated a manual reactor trip (RT) {JC} at 07:41 hours, as a result of lowering steam generator (SG) {AB} levels due to the loss of feedwater (FW) {BA} from the trip of both main boiler FW pumps (MBFP) {P}, and entered emergency operating procedure EOP 2-E-0 (Reactor Trip or Safety Injection). All control rods {AA} fully inserted and all required safety systems functioned properly. The plant was stabilized in hot standby with decay heat being removed by the main condenser {SG}. The Auxiliary Feedwater System {BA} automatically started as expected. There was no radiation release. The Emergency Diesel Generators {EK} did not start as offsite power remained available. No primary or secondary code safety valves lifted. The 23 and 24 Main Steam Isolation Valves (MSIVs) {JM} failed close as a result of the loss of Instrument Air (IA) {LD}. The 21 and 22 MSIVs remained open. Equipment that did not perform properly included 1) The 22 MBFP High Pressure stop valve (MS-6-2) failed to fully close, 2) The IA Compressors did not Auto start, 3) The 21 FW regulating valve (FCV-417) packing leaked, 4) 23 Hot Leg Temperature was observed to be low, 5) Abnormal backup nitrogen consumption. Received Nitrogen Makeup IA Low Pressure Auxiliary Boiler FW Pump alarm at 08:14 hours. The event was recorded in the Indian Point Energy Center corrective action program (CAP) as CR-IP2-2013-02717. A post trip evaluation was initiated and completed on July 3, 2013.

Prior to the event piping lines including the IA line buried in the main transformer moat were excavated and temporary supports installed. The work was performed under the direction of Maintenance Support. There was no work being performed on the IA line or its supports at the time of the event. Investigations determined the decreasing SG levels were due to a loss of main FW flow as a result of the closure of the FW regulating valves {FCV} (FCV-417, FCV-427, FCV-437, and FCV-447). The FW regulating valves closed due to a loss of IA pressure. The IA pressure was lost when a two inch IA copper tubing {TBG} in the 22 Main Transformer {EL} moat separated at a soldered coupling {CPLG}. The IA line runs underground from the Control Building {NA} to the Auxiliary Feedwater Building {NF} and the Intake Structure. IA is routed to the FW regulating valves contained within the Auxiliary Feedwater Building. The two MBFPs are located in the Turbine Building and FW pump discharge is routed to a common header through three high pressure heaters then the FW leaves the Turbine Building and crosses the pipe bridge to the Auxiliary FW Pump Building which contains four separate FW lines with main FW regulating valves for regulating FW flow to the SGs. The IA compressors, air receiver, filters, and air dryers are located on the ground floor of the Control Building. Between the Control Building and the Auxiliary Feedwater building the two inch IA line passes underground through the 22 Main Transformer moat. The IA line is two inch copper tubing, ASTM B-88, Type K, nominal thickness (0.083 inches) with wrought copper solder end fittings. The underground part of the IA line is coated with coal tar epoxy and fabric. The solder joints of the line in the moat area were made up during original plant construction. An inspection of the tubing and fitting showed that the tubing was not fully inserted into the fitting socket/coupling (approximately 1/3 inserted) resulting in having only about 1/3 of the joint's design strength. The soil and epoxy coating on the line provided additional axial restraint preventing its separation while it was buried. Excavation of the line removed the axial restraint provided by the soil. Some additional stress was applied to the joint when the continuous deadweight support of the soil was replaced by temporary deadweight supports placed every 6 to 8 feet under the excavated line.

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An extent of condition investigation determined that all of the pipes located inside the 22 Main transformer moat and impacted by the excavation are Carbon Steel (CS) Schedule 40 piping (3 and 8 inch SW piping, 1 and 8 inch City Water piping, 6 and 8 inch Fire Protection piping) except the 2 inch IA tubing. The 3, 6 and 8 inch CS piping was fabricated with full penetration butt welds and the 1 inch CS piping was fabricated with socket weld fittings. The 2 inch IA line is copper tubing fabricated with wrought copper soldered fittings. Carbon steel piping is not susceptible to the same failure mechanism which caused the failure of the 2 inch IA line copper tubing. The basis for this conclusion is the pipe material (steel vs copper) and welded versus soldered joints. Welded joints have stronger connections than soldered joints. However, other soldered joints in the 2 inch copper line could have the same structural condition as the failed joint. To locate other possible line weaknesses, a pressure test was performed and an acoustic emission test and snoop testing was performed on the repaired coupling and the upstream and downstream elbows to identify any leakage. Results of the testing were satisfactory. For IA tubing within buildings, the improper installation found in this event would not be likely as the condition would have already failed due to low design strength, limited axial restraints, system/building vibration, and stress due to span supports. For the underground IA it is continuously supported and axially restrained preventing tube/fitting separation.

Cause of Event

The apparent cause was poor legacy workmanship assembling the IA tubing coupling during original construction. The copper tubing was not fully inserted into the fitting socket (approximately 1/3 inserted) resulting in having only about 1/3 of the joint's design strength. The soil and epoxy coating on the line provided additional axial restraint preventing its separation while it was buried. Excavation of the line removed the axial restraint provided by the soil. Some additional stress was applied to the joint when the continuous deadweight support of the soil was replaced by temporary deadweight supports placed every 6 to 8 feet under the excavated line.

Corrective Actions

The following corrective actions have been performed under Entergy's Corrective Action Program to address the cause.and prevent recurrence:

- The 2 inch copper IA tubing was reassembled and soldered with full insertion of the tubing into the coupling joint in accordance with ANSI Standard B16.22 (Wrought Copper and Bronze Solder-Joint Pressure Fittings).
- Acoustic emission and snoop testing was performed on the repaired coupling and the upstream and downstream elbows to identify any leakage. No leakage was identified.
- Axial and thrust restraints were installed on the 2 inch IA line in the moat.
- A caution was placed in the Buried Piping Program IDDEAL database associated with buried copper tubing identifying the potential for the separation of soldered joints when the line is excavated and the need for restraints or other contingencies to minimize the probability of a line separation.

Event Analysis

The event is reportable under 10CFR50.73(a)(2)(iv)(A). The licensee shall report any event or condition that resulted in manual or automatic actuation of any of the systems listed under 10CFR50.73(a)(2)(iv)(B). Systems to which the requirements of 10CFR50.73(a)(2)(iv)(A) apply for this event include the Reactor Protection System (RPS) including RT, AFWS actuation, and actuation of multiple main steam isolation valves (MSIVs).

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NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

This event meets the reporting criteria because a manual RT was initiated at 07:41 hours, on July 3, 2013, and the AFWS actuated as a result of the RT. Two of four MSIVs closed (23 and 24 MSIVs) due to the loss of IA thereby qualifying as multiple MSIV actuations. On July 3, 2013, a 4-hour non-emergency notification was made to the NRC at 10:46 hours, for an actuation of the reactor protection system {JC} while critical and included an 8-hour notification under 10CFR50.72(b)(3)(iv)(A) for a valid actuation of the AFW System and closure of multiple MSIVs (Event Log #49171). As all primary safety systems functioned properly there was no safety system functional failure reportable under 10CFR50.73(a)(2)(v).

Past Similar Events

A review was performed of Licensee Event Reports (LERs) reporting a RT as a result of main FW reduction. The review identified LER-2009-002. LER-2009-002 reported a manual RT on April 3, 2009, due to decreasing SG levels caused by loss of the 21 Main FW pump and failure of the main turbine to automatically runback. The direct cause was failure of the autostop oil tubing/Swagelock fitting on the main FW autostop oil header. The root cause was improper tubing installation due to poor worker practices. This LER is similar as it involved tubing and a failure due to improper installation. The corrective actions for LER-2009-002 included replacement of the fractured tubing, reconfiguration to meet installation requirements and training on Swagelock fitting installation. The corrective actions for that event would not have prevented this event as the issue was with a Swagelock fitting and stainless steel tubing and was internal to equipment and not a support issue with excavated buried copper tubing soldered to a coupling.

Safety Significance

This event had no effect on the health and safety of the public.
There were no actual safety consequences for the event because the event was an uncomplicated reactor trip with no other transients or accidents. Required primary safety systems performed as designed when the RT was initiated. The AFWS actuation was an expected reaction as a result of low SG water level due to SG void fraction (shrink), which occurs after a RT and main steam back pressure as a result of the rapid reduction of steam flow due to turbine control valve closure.

There were no significant potential safety consequences of this event. Operators for this event anticipated a possible low SG level and actuated a manual RT. The manual actuating devices are independent of the automatic trip circuitry and are not subject to failures which make the automatic circuitry inoperable. There are two manual trip buttons, one located on flight panel FCF and the other on safeguards supervisory panel SBF2. Either one of these buttons will directly energize the trip coils of the reactor trip and bypass breakers in addition to de-energizing the undervoltage coils of the reactor trip and bypass breakers. The Reactor Protection System (RPS) is designed to actuate a RT for any anticipated combination of plant conditions to include low SG level. The reduction in SG level and RT is a condition for which the plant is analyzed. A low water level in the SGs initiates actuation of the AFWS. SG level instrumentation was available for a low SG level actuation which automatically initiates a RT and AFWS start providing an alternate source of FW. The AFW System has adequate redundancy to provide the minimum required flow assuming a single failure. The analysis of a loss of normal FW (UFSAR Section 14.1.9) shows that following a loss of normal FW, the AFWS is capable of removing the stored and residual heat plus reactor coolant pump waste heat thereby preventing either over pressurization of the RCS or loss of water from the reactor. For this event, rod control was in automatic and all rods inserted upon initiation of a RT. The AFWS actuated and provided required FW flow to the SGs. RCS pressure remained below the set point for pressurizer PORV or code safety valve operation and above the set point for automatic safety injection actuation. Following the RT, the plant was stabilized in hot standby.